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ENGINEERING PROBLEMS OF POWER REACTORS

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ENGINEERING PROBLEMS OF POWER REACTORS

PREFACE

This paper will illustrate some of the problems facing those who design and build nuclear power reactor systems. The field of nuclear power is somewhat unique in that engineering development is lagging the work of the research groups to an extent that the entire nuclear program has suffered. Many nuclear systems have been developed by the nuclear engineers and physicists, which in many cases improve plant operation and economy but which cannot be built practically.

In considering these problems this paper will be limited to heterogeneous reactors cooled by either water or liquid metal. Since both types have been or are now being built, this discussion will dwell on the general problems of each. The Pressurized Water Reactor will, in all probability, be the most common of the nuclear power reactors built within the next five years; therefore, special emphasis will be given to this type of system.

PRIMARY COOLANT

The first factor to consider is the primary coolant and its effect on the equipment design. Since this discussion will be a non-nuclear one, differentiation will not be made between H_2O and D_2O in an aqueous system. Sodium will be considered as a typical coolant when examining a liquid metal system. Sodium has been commonly used and is representative of all liquid metal coolants.

The following requirements are common to all reactor coolants:

1. The coolant must be a good heat transfer medium.
2. The coolant must be amenable to maintaining high purity.

3. The coolant must cause little or no corrosion of the fuel elements or primary system equipment.
4. The coolant must be contained. A high degree of leak tightness is a requirement of any reactor system.

These requirements impose various problems on the engineer designing a water or sodium system. Each requirement will now be considered for both types of system.

Good Heat Transfer Medium

Water has a high heat capacity but it also has a high film drop. Briefly, a film develops on the heating element. This film impedes heat flowing from the heating element to the water coolant. Probably the most serious problem imposed by a water coolant is the critical temperature limitation of water. In order to obtain a high enough coolant outlet temperature to be practical, the system must be capable of containing high pressures. In early reactor designs even localized boiling was considered hazardous. More recently some localized boiling has not been considered detrimental. However reactors are still being designed so that the maximum core temperature is somewhat less than the primary coolant saturation temperature.

Sodium has the advantages of virtually no film drop and a high boiling temperature. Sodium does react violently with water. Special precautions are necessary to insure the primary coolant's being isolated from the steam system. In general, there are two ways of isolating the radioactive sodium. One would be to design a steam generator so that a second tube will incase the tube in which the sodium is flowing. The volume between the tubes might be filled with an inert gas. The water would be outside of the second tube. Thus, if sodium leakage did occur,

the water could be protected by a secondary barrier. A second method would be to include an intermediate loop in the system. The radioactive sodium would transfer the heat in an intermediate heat exchanger to a less active fluid such as NaK. The NaK would then act as the heat-transferring agent in the steam generator.

Purity

There can be no in-leakage of impurities in the primary coolant, and provision must be made for a high purity make-up system. The reasons for this requirement are obvious. It is imperative that the fuel elements be free from foreign deposits. Certain impurities will accelerate corrosion. Most impurities will become highly radioactive, and if deposited on other equipment in the system, such as pumps and valves, they can create sizable maintenance problems.

High purity water can be achieved without too much difficulty. But, high purity water is a highly corrosive fluid. Therefore, the requirement of high purity in an aqueous system sets up a never-ending cycle. High purity causes corrosion. Corrosion requires a clean-up loop to maintain high purity. Water does have the advantage of being less radioactive than sodium. In general, pure water has only a seven-second half-life due to N^{16}. The impurities common to water, such as iron, nickel and manganese, do not have this advantage. For the most part, they become highly radioactive, have long half-lives, and must be removed. To sum up, in a water system a complete clean-up loop is necessary. The resultant purity must be two ppm maximum dissolved solids in a system.

The properties of sodium are such that the coolant becomes highly radioactive when it comes in contact with the nuclear core. Thus, the activity of the impurities in the coolant is not so important.

One of the greatest problems affecting purity in a sodium system is the tendency of free oxygen to form sodium oxide. It is undesirable to have a deposition of this material on the fuel elements and other equipment. Steps must be taken to remove this sodium oxide from the system. A method of removal is explained later.

Corrosion

As mentioned above, high purity water is very corrosive. Only zirconium, 300 series stainless steels and such metals as stellite, etc., are corrosion resistant, to the degree required. As yet no good corrosion inhibitor is available for a high temperature water system. Various techniques have been used to limit corrosion. It has been found necessary to maintain an excess of hydrogen in the system. This excess of hydrogen is used to combine with and eliminate excess oxygen. To give an example of the importance of this, numerous tests have been run on various mechanisms operating in hot water. In the Argonne National Laboratory, a rack and pinion was operated in water at 450° with an excess of oxygen. The material of the rack and pinion was a 400 series stainless steel. After fifteen hours of operation the rack and pinion galled and seized. A second series of tests was run involving identical components operated in identical conditions with the exception that an excess of hydrogen was maintained in the system. The rack and pinion ran for five million cycles, and upon examination only a slight polishing of the gear teeth was noticeable.

A recent development, still in the laboratory stage, has been the use of the element technicium as a water corrosion inhibitor. This element, which is a fission product of uranium, was very effective at the lower temperatures below 400° F.

Preliminary tests have indicated that the addition of ammonia in a hot water system is helpful in reducing corrosion. Since only several ppm of ammonia is used, any activity from resultant free nitrogen is not harmful.

It was mentioned above that the sodium oxide tends to form in the sodium coolant system. In addition to its other undesirable qualities, this oxide is highly corrosive. A method has been developed to remove sodium oxide from the system. The oxide freezes at a higher temperature than the sodium coolant. Cold traps can be incorporated in the system and the frozen sodium oxide can be intermittently removed. At temperatures above 1000° F. the sodium coolant itself becomes corrosive. The coolant temperature in a sodium system is usually maintained between 800° and 900° F.

Leakage Containment

The primary coolant pressure of a water system is usually from 1500 to 2000 psi. It has been the practice in earlier power reactor development to specify a zero leakage requirement of the primary coolant. There are certainly valid reasons for this requirement in a mobile reactor plant. It is felt, however, that this zero leakage philosophy should be revised when designing pressurized water systems for land-based power stations. For years central power stations have tolerated some nominal leakage in the system. It may well be that major cost reductions can be realized if nuclear power plant designers would follow the conventional power plant leakage requirements.

The problem still remains, even with some allowable leakage, of building equipment to withstand a pressure of 2000 psi. These problems

will be illustrated in more detail later when the individual components are discussed.

A sodium system is designed to operate at much lower pressures, in the range of 100 to 300 psi. The rigid leakage requirements which may be relaxed in the water system must be maintained in the sodium system. In addition to the requirements of structural integrity, sodium system components must have high metallurgical requirements. The radioactive sodium can and does penetrate inclusions on the surface of the components. Such a penetration would be very difficult to remove, and it could cause a violent reaction. The sodium coolant has a 14.8 hour half-life and certain isotopes such as Na_{24} are extremely active. Leakage of this isotope would be very hazardous. In designing a sodium system, the engineer must still consider zero leakage to be a firm requirement.

There is a factor to consider when a comparison is made between a water and sodium coolant. This factor is the reactor moderator. The function of a moderator is to slow down the rate of travel of the emitted neutrons so that their availability for capture will be greater. In a thermal reactor, the neutrons are slowed down quite appreciably. They move somewhat faster in an intermediate reactor and a good deal faster in a fast reactor. Water has the property of being both a coolant and a moderator. It is almost unique in this property. Most sodium coolants are used in intermediate and fast reactors. In an intermediate reactor, a moderator must be used in addition to the sodium. A typical moderator might be beryllium. This necessity of a special moderator certainly has an effect on an economic comparison of the two types.

REACTOR SYSTEM EQUIPMENT

Fuel Elements

A high fuel density is required in reactor engineering because of the high fuel cost and fuel element fabrication and reprocessing costs. The table shown below will give an example of the comparative heat fluxes in a coal fired boiler to those in a reactor core.

COAL FIRED BOILERS[1]

Maximum in Burner Area -	100,000 Btu/hr ft^2	to	200,000 Btu/hr ft^2
Average -	35,000 Btu/hr ft^2	to	50,000 Btu/hr ft^2

REACTOR CORE

Maximum Core Center Line-	150,000 Btu/hr ft^2	to	300,000 Btu/hr ft^2
Average -	50,000 Btu/hr ft^2	to	100,000 Btu/hr ft^2

The fuel elements used in an aqueous system must be clad with a non-corroding material. If an unclad uranium fuel element was used in a water system, the corrosion of the uranium would be measured by the yard. The corrosion problem when sodium contacts uranium is not as serious. In future reactor designs, fuel element cladding may not be required with a sodium coolant. The necessity of cladding the aqueous system fuel elements imposes a fabrication problem on the reactor designer and builder. The materials to be clad, such as uranium and plutonium, as well as certain cladding materials, such as zirconium, are relatively new materials, and a backlog of fabrication experience is not available. In order to illustrate the fabrication problems of

fuel elements it may be well to go through a typical fabrication procedure of a hypothetical fuel element. The fuel element may be an assembly of fuel plates. The first step is to obtain a thin, rectangular sheet of uranium. This slab of uranium may then be placed between two oversized slabs of cladding material, such as zirconium. In effect we now have a sandwich, the uranium being the meat of the sandwich. This sandwich can now be rolled and the outside of the cladding welded or brazed. The welding or brazing is necessary in order to render the plate leak-proof. During reactor operation the fuel will generate fission products in the form of gases. These products are highly radioactive. The fuel element cladding must contain these fission products.

A fuel assembly may contain several fuel plates. These plates can fit into two grooved side sections. The grooves must be machined so that the spacing between the plates will be consistent. The bottom and top of the assembly is left open so that the coolant will flow through the spaces between the plates. It is extremely important that close tolerances be held on the spacing grooves aligning the fuel plates. If the spacing is not consistent, the predetermined mass of coolant will not flow by a certain fuel plate. The fuel plate, therefore, could overheat and possibly melt the cladding. The gaseous fission products would then be carried throughout the system and would be deposited on the primary loop components.

There are other types of well-known fuel elements, such as rods and cans. The fabrication processes are similar in that a consistent cladding and spacing is absolutely necessary.

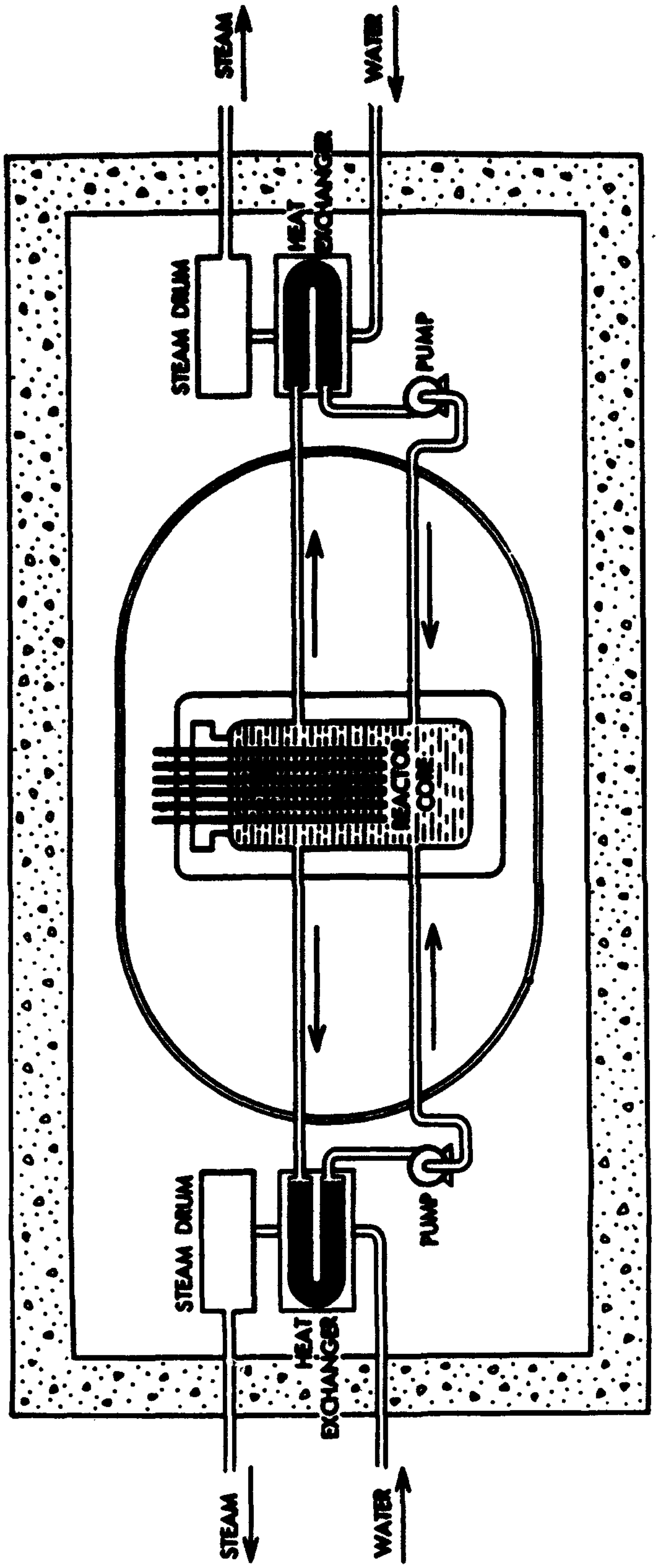

STEAM
WATER
HEAT EXCHANGER
STEAM DRUM
PUMP
REACTOR CORE
PUMP
STEAM DRUM
HEAT EXCHANGER
STEAM
WATER

Figure # 1

Reactor Core Components

In addition to fuel elements a typical reactor core would contain the following components:

1. Flow guides to insure an equal flow distribution throughout the core.
2. Control rods to control the reactor.
3. Bearings for these rods as they move.
4. A thermal shield to alleviate thermal stress in the reactor vessel.
5. A hold down device to prevent the fuel elements from moving or chattering under high flow conditions.

All of the above components must be fabricated from a special non-corrosive material. Since physical positioning must be maintained with extreme accuracy in a reactor core, all the components must be fabricated to close tolerances. The rod bearings cannot be lubricated and yet the rods must have complete freedom of motion. The hold down device must be capable of giving core rigidity and yet must be a quick opening device for ease of core reloading. These problems are typical in that they are relatively complex and interrelated.

Control Systems

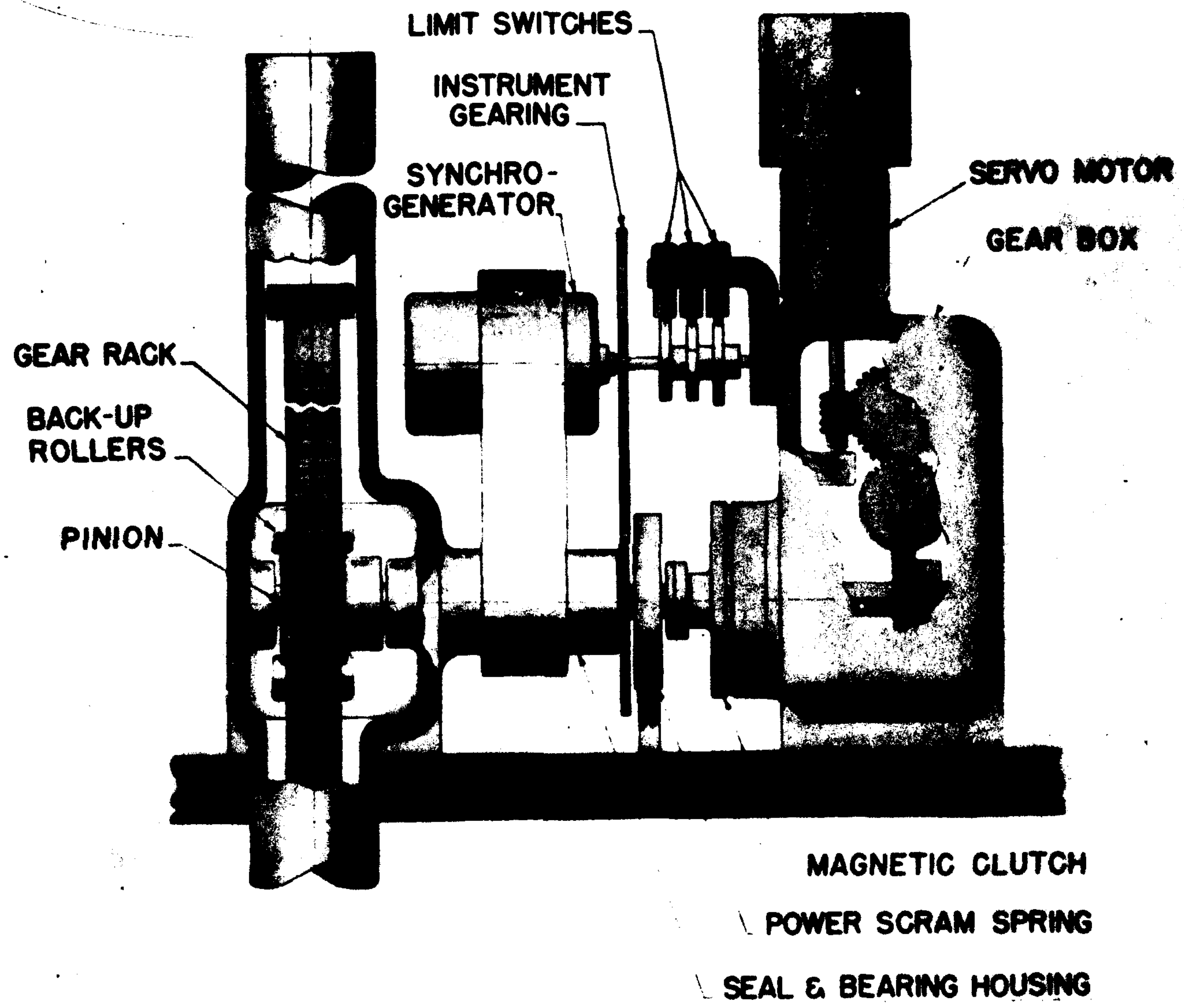
A general philosophy of control is to use as a control point a fixed reactor outlet temperature. Figure 1 illustrates the usual double loop used in a pressurized water system. If an increased demand appears at the turbine shaft, the turbine throttle opens, the pressure in the line to the turbine falls, more heat is transferred from the primary loop to the secondary steam system, and eventually the temperature out of the reactor, assuming no increase in the return of heat generation there, will

fall. This creates an error signal to the control system, which causes the power level of the reactor to be raised to meet the increased demand. A decrease in power demand results in the reverse condition.

The reactor power level can be controlled by the insertion or removal of control rods. The rods must be made of a material that is, in effect, poisonous to the nuclear reaction. Three typical materials that might be used would be cadmium, boron, or hafnium. Since cadmium also corrodes in water, it too must be clad. It has the advantage of welding easily, but it is not usually used in a sodium system since it melts at the higher operating temperatures. Boron is brittle and is used only in such forms as boron steel or boron carbide. Hafnium is expensive but does have good corrosion resistance, good physical properties, and welds easily.

As mentioned above, the motion of the rods will be controlled by the plant output requirements. There are emergency conditions which will override the normal control and cause an extremely fast insertion of all rods. This fast insertion is commonly called a scram. There may be any number of events, depending upon reactor design, which would cause a scram. An extremely fast rise of the primary coolant outlet temperature might be such an event. A rise in the radiation level as detected by appropriate nuclear instrumentation would most certainly cause a scram.

It is sometimes required to add an extension of stainless steel to a control rod, thus making it longer and more unwieldy. When a control rod is withdrawn from the reactor core, a void will be left in the core. If water were to fill this void, neutron flux peaking would result in this area. The extension, therefore, would fill the void with steel rather than with water. This extension will impose problems to the



CONTROL ROD DRIVE

Figure # 2

designer of the flow guide. The guides must be positioned so that the control rod extension will have clearance when the rods are entirely in the reactor core.

When control rods are used they must be driven by a control rod drive mechanism. There are four general requirements of a control rod drive mechanism:

1. The mechanism must induce reciprocating or rotary motion.
2. The mechanism must be capable of fineness in control. In effect, when the operator asks the rod to stop moving, the rod must stop moving with as little coasting as possible.
3. The mechanism must indicate the position of the rod to the operator at all times.
4. The mechanism must be capable of reinserting (scramming) the rod into the reactor core with an acceleration of approximately 1 G.

The illustration below shows a simplified drive mechanism. It will be noticed that the mechanism employs a rotary shaft seal. In using such a seal some leakage must be expected. If this leakage can be collected, it is felt that such a mechanism would be acceptable from a plant operation standpoint, and certainly would be more economical than most drive mechanisms now in use.

As illustrated, the drive mechanism consists of a low-inertia driving motor, a reduction gear train, a quick releasing magnetic clutch, a scram spring, a simplified indication system, the rotary seal, a rack and pinion, and a rod latching device.

The low-inertia motor will be specified so as to achieve the requirement of fineness of control as well as inducing rod motion. The gear train is necessary to give the required slowness of rod travel. The magnetic clutch will quickly release the rod upon power failure or electronic impulse. The spring will then drive the rod into the core with the required acceleration. The instrumentation system consists of a gear-driven synchro-generator to fulfill the requirement of position indication and adequate limit switches to control rod travel.

The latch mechanism offers a complex problem. This mechanism must be the connection between the rod and the rack. It must do more than that, however. During reactor refueling the control rods must remain in the reactor core while the fuel elements are being replaced. The latch must be capable of being remotely disengaged from the rod so that refueling can be accomplished.

The rod mechanism described above typifies an economical device that may be used in an aqueous system. There have been many other such devices designed for both sodium and water reactors. Some of these devices are designed with a zero leakage requirement. Most of the mechanism, therefore, must be submerged in the coolant. The corrosion and lubrication problems alone impose stringent design requirements and do not aid in any way in reducing plant cost.

There is a factor to be considered which is typical to a water reactor and which greatly eases the problem of reactor control. This factor is known as a negative temperature co-efficient. In effect, as the coolant temperature rises, the effective neutron flux goes down. A very oversimplified explanation of this is that as the liquid expands or becomes less dense, all the nuclei of the coolant are pushed further apart, leaving more gaps through which neutrons may escape without causing fission.

Reactor Vessels

When designing a nuclear power plant the physicist desires the largest possible core volume in order to obtain a maximum power output. The design engineer again must be the limiting factor in core sizing, especially in an aqueous system. The reactor vessel now being constructed for the Pressurized Water Reactor to be installed in Shippingsport, Pennsylvania, will be one of the largest vessels ever built to withstand the required operating pressure. The design pressure of the vessel illustrated in Figure 3 is 2500 psi; the design temperature is 600°F. The inside diameter of the vessel is approximately nine feet. The vessel wall thickness is approximately eight and one half inches. Obtaining high quality plate for the vessel wall is very difficult. At this time it is felt that this plate thickness is maximum, since larger

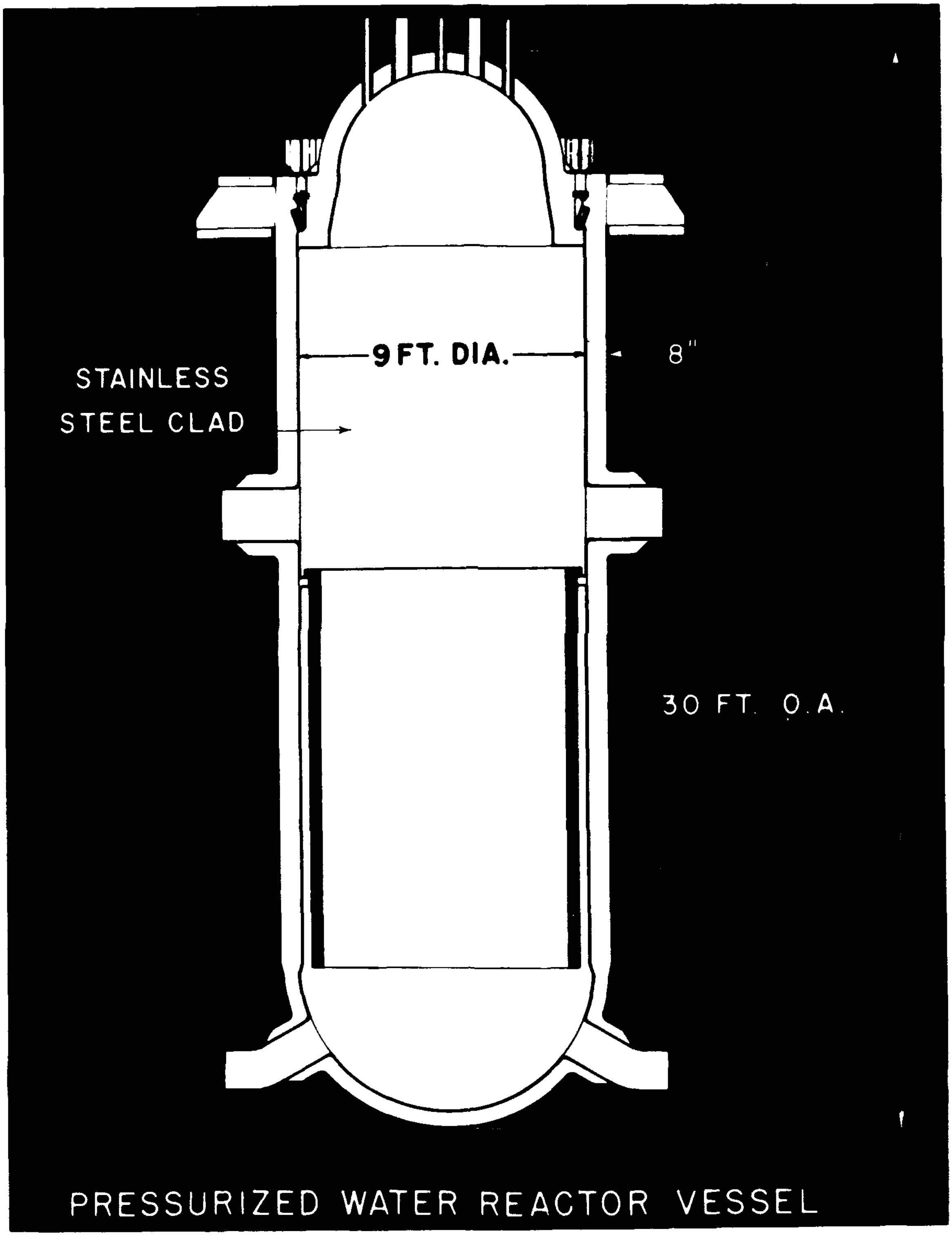

9 FT. DIA.
8"
STAINLESS
STEEL CLAD
30 FT. O.A.
PRESSURIZED WATER REACTOR VESSEL

Figure # 3

plates of high quality cannot, as yet, be guaranteed by the steel mills. Therefore we have reached the limit of vessel diameter for a 2500 psi type pressure. Should the nuclear physicist design a larger core the design engineer must consider multiplicity of vessels or a hitherto undiscovered vessel design.

In order to install and possibly reload the reactor, this extremely large vessel must be equipped with a removable head. The force required to hold down a head of nine feet diameter under a pressure of 2500 psi is approximately 25,000,000 pounds. Should bolts be used, each bolt would have a diameter of slightly more than six inches and approximately forty-five would be needed. It can be readily seen that if one utilises bolts one does not have a "quick opening" vessel cover. There have been other vessel cover hold down devices designed; one is illustrated in Figure 3. This closure is an adaptation of the well-known shear-ring design. It consists of a number of conical segments which are dropped between the reactor cover and vessel wall and then rotated so as to form a compression ring between the vessel wall and cover. This design does require some development, but it is felt it offers some distinct advantages over the more common bolted closure. These advantages include a shorter required removal time, smaller closure components and improved load distribution.

Another problem in the vessel fabrication is that only a non-corrosive material must contact the coolant. Therefore all internal surfaces of the vessel must be clad with stainless steel.

The reactor vessel in a sodium system is not as critical, since the operating pressures are much lower. This vessel does, however, have its own unique problems. It is common practice to design this vessel with a double wall so that a safety barrier is provided for sodium leakage. It

is also required to include electrical heaters in the assembly of the vessel so that the sodium can remain molten when the reactor is shut down.

Refueling

In the field of reactor design one phase of the process has received much less development than any other. This is the refueling of the nuclear core. Most nuclear system designs which have emanated from the National Laboratories have either overlooked or minimized the problems involved in stoking the nuclear furnace. Most of the research and test reactors are refueled manually. The equipment is cheap and time is not usually a factor. In a central power station the length of time that a power-producing unit is shut down must be kept to a minimum. Manpower is expensive. The problems encountered in developing a fast, economical reloading system are numerous. It may be well to outline a typical reloading scheme for a Pressurized Water Reactor and then show possible methods of eliminating certain steps and reducing the required shut down time.

The reactor is usually buried in a concrete chamber. For reasons which will be covered in the next section the reactor vessel is enclosed in a larger pressure vessel. There usually is a concrete shield plug covering the larger vessel. The following steps are required to refuel the reactor after it is shut down:

1. The concrete plug must be removed and stored.
2. The cover of the large vessel must be unfastened, removed and stored.
3. The hold down device of the reactor vessel cover must be removed and stored.

4. The reactor pit must be filled with water.
5. The reactor vessel cover must be removed.
6. The core hold down mechanism must be unlatched or removed, depending upon its design.
7. Fuel elements are then removed individually from the reactor core and placed in an underwater storage pit.
8. If necessary, the control elements are removed and placed in an underwater storage pit.
9. New control rods are lowered through the water shield into the core.
10. New fuel elements are lowered through the water shield into the core.
11. The core hold down device is replaced.
12. The reactor vessel cover is lowered into place.
13. The shielding water is drained from the reactor pit.
14. The reactor vessel cover closure is replaced and fastened.
15. The outer vessel cover is replaced and fastened.
16. The concrete plugs are replaced.

To illustrate the time required to perform such an operation consider Step Three above. Should the vessel cover be held down by bolts they must be heated with special elements in order to elongate them and relieve tension before they can be unfastened. Fortunately, there probably will be enough steel and water between the bolt head and the reactor core so that this device can be guided and operated by personnel in close proximity to the bolts. Let us further assume that it has been necessary to seal weld the reactor cover to the reactor vessel. A special cutting machine must be installed and must cut away the required weld metal. Considering the fact that this welded seam is forty feet in circumference,

it can be readily seen that this operation will be a time-consuming one.

In light of the above it is evident that rigorous development must be given to a more practical and speedier unloading system. This is especially true in a reactor design where the integrity of the fuel elements is doubtful and a fuel element rupture is not inconceivable. A ruptured element must be removed as quickly as possible.

If a device could be developed that would speedily refuel the reactor without removing the reactor vessel cover, a large step would have been taken in the practicability of Pressurized Water Reactors.

A sodium system does not have the problems of high pressure and rugged, heavy vessel closures. The sodium coolant does, however, create problems. There may be difficulty encountered in removing fuel elements directly from sodium into the atmosphere. Sodium must also be completely sealed during reactor operations. Several schemes have been developed to refuel sodium reactors. One typical scheme would be to incorporate a pool of sodium above the actual reactor vessel and lift the element into this pool and then transport the element to a canal. This canal would in all probability be filled with a sodium gas. Access to the elements might be provided by eccentric plugs which can be rotated when the reactor is shut down, so as to give clearance ports through which unloading mechanisms can operate.

In the sodium system designs to date, as well as the Pressurized Water Systems, refueling procedures have been unwieldy and too time-consuming for practical central power station operation.

Pumps and Valves

The extreme low leakage requirement of reactor systems has been the greatest problem to the designer of the primary system circulating pumps

and valves. Several pumps have been developed which will operate with zero leakage of the primary coolant. In most of these designs the rotating parts of the pump are totally immersed in the coolant. This type of pump is commonly called a canned rotor pump. The stator, or rotation inducer, is on the outside of the can and is externally cooled. To be consistent with the water system leakage philosophy discussed earlier, it may be possible to use a circulating pump with some nominal amount of leakage, provided that leakage is consistent and controlled.

At the present time most reactor systems must be designed with canned pumps. These pumps are very expensive and as yet those sized over ten thousand gpm are developmental.

A leak-proof quick opening valve is more difficult to design. At this time almost any leak-proof valve of the size required in a central station plant is a developmental item.

The problems above are encountered to a greater degree in a sodium system. The development of the electromagnetic pump has been an effective solution to the pumping problem of a liquid metal system. An electromagnetic pump is expensive, extremely large and comparatively inefficient. Some reactor designs have considered using gas-sealed sump type circulating pumps. The seal development alone has been going on for years, but as yet a satisfactory solution has not been found. The valve problems are greater in a sodium system due to the deposition of sodium oxide on the moving parts, which cause valve operation to become uncertain.

<u>Accident Containment</u>

One other problem, at least for the time being, confronts the reactor system designer. Safety is a most important consideration of any proposed nuclear power plant. Each proposed plant must be reviewed by a Reactor Safeguards Committee. When a reactor plant operator submits his

design for approval, he must also submit his definition of the "maximum credible accident" which might occur during plant operation. He then must provide means in the overall design for containing such an accident so that the safety of any surrounding area is insured.

A typical maximum accident of a Pressurized Water System would be that the reactor vessel or any portion of the primary loop ruptures during operation. The primary coolant would flash to steam and the steam would be radioactive. This steam must be contained. This is the reason for the larger pressure vessel mentioned during the refueling procedure. To contain such an accident the reactor primary loop and steam generator must all be contained in pressure tight vessels. Remembering that the reactor vessel has an outer diameter of approximately 11 feet and is approximately 35 feet high, one can imagine the size of a containment vessel. The pressure which must be contained will, of course, depend upon the volume of the vessel and the amount of water which has been calculated to flash into steam. Once the steam has been contained a controlled vapor release system must be included in the plant design.

CONCLUSION

This paper has illustrated some of the problems encountered by the nuclear power plant engineer and designer. Some of these problems are very difficult but none are insurmountable. Nuclear power plants have been built and have operated very successfully. They have proven to be stable, safe and efficient. To the engineer remains the problem of developing practical components that will render these plants economical for wide-spread central power application.